NON-PUBLIC?: N
ACCESSION #: 8905020429
LICENSEE EVENT REPORT (LER)

FACILITY NAME: McGuire Nuclear Station, Unit 1 PAGE: 1 of 18

DOCKET NUMBER: 05000369

TITLE: A Steam Generator Tube Rupture Occurred On March 7, 1989 and Resulted in an Alert Being Declared and an Unplanned Release of Radioactivity
EVENT DATE: 03/07/89 LER #: 89-004-00 REPORT DATE: 04/06/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(i), 50.73(a)(2)(ii), 50.73(a)(2)(iv), Other - Part 21

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: AB COMPONENT: HX MANUFACTURER: W120
REPORTABLE TO NPRDS: Yes

SUPPLEMENTAL REPORT EXPECTED: Yes EXPECTED SUBMISSION DATE: 09/01/89

ABSTRACT:

On March 7, 1989 at 2345, an Alert condition was declared on Unit 1 because of indication of primary to secondary leakage in excess of 50 gpm. At 2346, Unit 1 was manually tripped from Mode 1, Power Operation, approximately 83% power because of indications of a steam generator tube leak in the B Steam Generator. All major plant equipment functioned properly, and the unit was stabilized in accordance with Station Abnormal Operating Procedures. The initial notifications to the State, local county authorities and the resident NRC inspector were made at 2358. The NRC Region II Office was notified at 0044 on March 8, 1989. The Technical Support Center and Operational Support Center were fully staffed and activated by 0132.The Crisis Management Center was staffed and activated at 0740. The event is assigned a cause of Other because of the rupture of a single tube in B steam generator. The results of the inspection and analysis of the tube failure will be addressed in an addendum to this LER.

END OF ABSTRACT

EVALUATION:

Background

The Reactor Coolant system EIIS:AB! tran
ports heated water from the reactor
EIIS:RCT! to the steam generators (S/Gs) EIIS:SG!, where heat is transferred to the Feedwater EIIS:SJ! and Main Steam systems EIIS:SB!. The four S/Gs are Westinghouse Model D-2 vertical shell, U-tube evaporators EIIS:EVP!. There are approximately 4700 tubes in each S/G. Reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the channel head of the S/G. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. The heat transfer tubes and the divider plate are made of Inconel and the interior surfaces of the reactor coolant channel heads and nozzles are clad with austenitic stainless steel. The primary side of the tube sheet is weld clad with Inconel.

All volatile treatment chemistry has always been used on Unit 1. (All volatile treatment is a method of pH and oxygen control using hydrazine and ammonia hydroxide as the principle chemical control agents.) Unit 1 has been operated well within the Westinghouse recommendations for use of boric acid treatment on the primary system since 1979. The EPRI guidelines for primary and secondary water systems have been followed for the past three years. The station has met the intent of the guidelines and has met the majority of the actual numerical limits. The last cycle, with respect to out of guideline conditions on Unit 1, was one of the best from a chemistry standpoint that it has ever been. However, higher than normal chemical hideout return was seen compared to the previous Unit 1 Refueling Outage.

Previously, 197 tubes had been plugged ia the B S/G. The plugging history of BS/G is similar to the plugging histories of S/G A, C, and D. There is no information in the inspection/repair history which would indicate that the B S/G is any more susceptible to tube degradation than any of the other S/Gs.

Eddy current testing of 100% of the tubes in the hot leg of the S/Gs is performed during each Refueling Outage. The hot leg is much more susceptible to Primary Water Stress Corrosion than the cold leg primarily because of elevated temperatures. During the last Refueling Outage in the winter of 1988, some eddy current testing was performed in the cold legs of the S/Gs. All of the tubes in the A & C S/G cold legs were partial length eddy current tested. Twenty percent of the tubes in all four S/G cold legs were full length eddy current tested during this outage.

The region of the defective tube was tested during preservice inspection. No further testing had been done since then. The data for the preservice inspection was taken with a single frequency machine and did not provide the information that is available using current testing techniques.

Unit 1 achieved commercial operation in December, 1981 and has completed operation of its fifth fuel cycle.

Description of Event

Prior to the tube rupture, Unit 1 was operating at 100% power after being returned to service following a Refueling Outage on December 31, 1988. Reactor coolant and secondary system conditions were normal. Reactor coolant system leakage calculated on March 6, 1989 resulted in 0.377 gpm identified leakage and 0.199 gpm unidentified leakage. A primary to secondary leak on B S/G had been detected in January, 1989 and was being monitored. The leak was approximately 10-15 gallons per day. Operation of charging and letdown was normal with the B Centrifugal Charging Pump EIIS:P! in service. The Main Steam Power Operated Relief Valve EIIS:V! for B S/G was isolated for valve packing repairs.

On March 7, 1989 at 2338, 1EMF-25, Steam Line B Radiation Monitor EIIS:IL!, alarmed and would not reset. The Control Room EIIS:NA! Operators observed B S/G feedwater flow decreasing while B S/G narrow range level indication remained - relatively constant. In addition, Pressurizer EIIS:PZR! level was decreasing. The Control Room Operators immediately recognized this incident as a S/G tube leak and implemented procedure AP/1/A/5500/10, NC System Leakage Within the Capacity of Both NV Pumps - Case I - Steam Generator Tube Leakage. The next indication was when 1EMF-33,Condensate Air Ejector EIIS:WF! Radiation Monitor, alarmed. The Control Room Operators initiated a 30 Mwe/min load reduction and started the A Centrifugal Charging Pump. In addition, the Control Room Operators reduced letdown flow from 75 gpm to 45 gpm. To compensate for the load reduction, emergency boration was initiated. This resulted in some fluctuation in Reactor Coolant T-ave causing fluctuations in Pressurizer pressure and level. (Reference page 13 of 18).
At this point, 1EMF-36(L), Unit Vent EIIS:VL! Low Range Radiation Monitor, alarmed.

At 2345, Control Room Operators declared an Alert based on indications of primary to secondary leakage. At 2346, the Control Room Operators initiated a manual Reactor Trip which resulted in an automatic Turbine Trip. At this time, primary to secondary leakage was estimated to be 100-150 gpm based on flow to the cold legs and mismatch flow from charging to letdown. Procedure

AP/1/A/5500/01, Reactor Trip, was implemented. At the time of the Reactor Trip Pressurizer level indication on Control Room instrumentation was approximately 36 - 38%. To aid in restoring Pressurizer level, the Control Room Operators opened valves 1NI-9 and 1NI-10, Boron Injection Tank Discharge Isolation, and swapped the suction of the Centrifugal Charging Pumps to the Refueling Water System EIIS:DA! Storage Tank (FWST). In addition, the Control Room Operators began immediately to isolate B S/G and initiate Reactor Coolant system cooldown and depressurization. By 0025, the pressure in B S/G and in the Reactor Coolant system was essentially equalized. All primary systems responded satisfactorily and all parameters were controlled satisfactorily. The NC System Leakage Within the Capacity of the NV Pumps procedure directed the Control Room Operators to implement procedure OP/1/A/6100/02, Controlling Procedure for Unit Shutdown. In accordance with the shutdown procedure, the Control Room Operators blocked the actuation circuit for Safety Injection to prevent an unnecessary automatic actuation. The Control Room Operators stated that when Safety Injection was blocked they knew that the plant had been stabilized and that the leak was under control. In addition, during the transient, Pressurizer level indication in the Control Room did not go below 10%.

If the Pressurizer level indication had dropped to 5%, the Control Room Operators would have manually initiated Safety Injection in accordance with procedure.

An unplanned radioactive release occurred as a result of the S/G tube rupture. The release resulted from a steam release from the A, C, and D S/G Power operated Relief Valves and from the Condensate Steam Air Ejector discharge to the unit vent. The total unplanned release averaged over the two hour period from 2340 on March 7, 1989 to 0140 on March 8, 1989 was 12.72 Curies (Xenon-133 equivalent) and 5.24 E-4 Curies (Iodine-131 equivalent).

At 0322, on March 8, 1989, further cooldown of the Reactor Coolant system began. Blowdown to the condensate polishers from B S/G was initiated at approximately 0312 and approximately 0410 and secured at approximately 0345 and approximately 0543, respectively, to control the level in the S/G. (Reference page 14 of 18.) Both times that Blowdown occurred from B S/G,an increase of activity at the unit vent resulted. Health Physics personnel expected these increases in activity. Activity in the unit vent from 0310 until 0410 and from 0420 until 0520 indicated that the concentration 2 at the site boundary was 0.75 MPC and 0.70 MPC, respectively. The Technical Specification limit of 2 MPC at the site boundary averaged over these one hour intervals was never exceeded. Offsite dose monitoring teams were dispatched at 0640. Twelve environmental samples, and two iodine and particulate samples were obtained and analyzed. All results were equivalent, to background activity. One environmental air sample located at the site boundary, .5 miles SW, was collected between March 7, 1989 and March 14, 1989. This sample, for the first time, had results above background. The sample results were 2E-14 uCi/ml Iodine - 131. This value is only 2% of the Technical Specification reporting fraction and is, therefore, not reportable.

At approximately 0705, the leakage assessment was changed from the original estimate of approximately 100 - 150 gpm. The Reactor Engineer began reviewing transient monitor data between 0500-0600. He performed a Reactor Coolant system leakage calculation based solely on Pressurizer level change prior to the Reactor Trip. He estimated the tube rupture leakage averaged approximately 540 gpm. His estimate did not take into consideration any other volume changes (letdown, charging, etc).

At 1015, the cooldown of B S/G was started using the backfill method described in procedure EP/1/A/5000/04, Steam Generator Tube Rupture. Mode 4, Hot Shutdown, was entered at 1025. At 1640, both trains of the Residual Heat Removal system EIIS:BP! were in service. Unit 1 entered Mode 5, Cold Shutdown, at 1744. The Alert was terminated at 1815.

The initial notifications of the Alert condition were made to the state of North Carolina and local counties at 2358, on March 7, 1989, in accordance with procedure RP/0/A/5700/02, Alert. Hourly follow-up notifications were made to the state and local counties until the Alert was terminated. The NRC Resident Inspector was also notified at 2358 on March 7, 1989. The NRC Region II office was notified at 0044 on March 8, 1989, in accordance with procedure RP/0/A/5700/10, NRC Immediate Notification Requirements. The Technical Support Center and Operational Support

Center were both fully staffed and activated by 0132 on March 8, 1989. The Crisis Management Center EIIS:NC! was staffed and activated at 0740 on March 8, 1989.

A Steam Generator Tube Rupture (SGTR) Recovery Group was formed to determine the cause of the tube rupture and to develop and implement a recovery program. The recovery group organization includes representatives from station and General Office groups. In addition, a Technical Review Committee comprised of recognized Steam Generator industry experts was set up to provide an independent technical overview of the station's recovery effort. This committee consists of experts from Westinghouse, Babcock and Wilcox, Commonwealth Edison, Dominion Engineering, and an independent nondestructive examination consultant.

Conclusion

This event was assigned a cause of Other because of the rupture of a single tube in B S/G. The defective tube was identified as tube 18-25 during visual inspections conducted on March 14, 1989. The defective tube was leaking from the cold leg side. The defect is an axial defect approximately 3.48 inches in length and .2 inches wide at the maximum rupture opening. (Reference pages 15 and 16 of 18.) The defect was approximately 28 inches above the tube sheet. The defect started below the 20th tube support plate and ran approximately 1/4 inch above the support plate. Currently, the leading probable cause of the tube failure is corrosion assisted cracking of some type. A final report will be issued documenting the results of the investigation and failure analysis of the defective tube.

Plans for addressing the tube failure are currently being implemented. Work inside the S/Gs is anticipated to be completed by approximately April 23, 1989. The plans address four major areas: Eddy Current Testing, Visual Inspection, Tube Removal, and Metallurgical Examination. Eddy current testing of 100% of the tubes using a bobbin coil was conducted on all 4 S/Gs. A rotating probe inspection of the tubes around tube 18-25 in the B and D S/Gs was performed. In addition, rotating probe inspections were conducted on selected tubes in all 4 S/Gs. Tubes that have exhibited previous defects, new defects, and some tubes from the same heat (same manufacturing group) as tube 18-25 were included in this inspection. Eddy current testing was performed to determine the orientation of the defect in the tube and profileometry of the defect will be done. Visual inspections were conducted through the hole of an adjacent tube that was pulled. The tube pull operation removed tube 19-24 which is adjacent to the defective tube and then removed the defective tube itself. In addition, tube 13-34 was removed because of a long axial d
fect. Metallurgical analysis is in progress on tubes 18-25 and 13-34. Mockup testing conducted by Babcock and Wilcox indicated that a tube with defects similar to the tube 18-25 defect could be removed using available techniques. Engineering analysis will be conducted in several areas. Rupture mechanics will be evaluated. The actual versus the theoretical leak rate will be analyzed. The local thermal hydraulics conditions in the 20th tube support plate area will be evaluated. An evaluation will be done to analyze possible foreign object wear rate and subsequent defect propagation. The properties of the material heat for tube 18-25 will be evaluated. This scope of work is based on an evaluation of potential failure mechanisms and should provide a basis for justifying future S/G integrity.

A best estimate calculation of primary to secondary leak flow rate and integrated leakage was performed by the Safety Analysis Group of Design

Engineering. Initial leak flow was approximately 500 gpm, and the integrated leakage prior to primary/secondary pressure equalization was approximately 16,000 gallons. When backfill of B S/G was initiated, the best estimate of integrated leakage totaled approximately 67,000 gallons.

The Control Room Operators immediately recognized this event as a steam generator tube rupture. The Control Room Operators felt that the procedures they used gave good guidance. They also praised requalification training and simulator training and felt that the training had prepared them to handle the event. The Control Room Operators responded to the transient and stabilized the unit in a timely manner. The Control Room Operators used two procedures, Controlling Procedure for Unit Shutdown and NC System Leakage Within the Capacity of the Nv Pumps, at one time, to control Reactor Coolant pressure. Operations personnel will review these - procedures and make appropriate changes to improve the structure of the procedures. The Control Room Operators referred to the Emergency Procedure for a steam generator tube rupture, but were never actually using it because the criteria to implement the procedure were never met.

The Technical Support Center (TSC) and the Operational Support Center provided guidance and assistance to the Control Room Operators. The TSC was delayed in being fully activated because of freezing rain and ice buildup on the roads, but it was fully activated within one hour and forty five minutes. The responsiveness and benefit of emergency drills and training were evident in the actions of the members of the TSC and Operational Support Center. The Crisis Management Center accepted turnover for their key functions and provided additional expertise and guidance in the handling of the event.

The prevailing attitude of personnel in the TSC throughout the event was one of caution. This attitude reflected a methodical, thorough, and analytical approach to achieving cooldown and depressurization. The TSC personnel deliberately chose this controlled approach for the following reasons:

o the unit was stable

o the Control Room Operators had good control of the event

o radiation conditions were known

o fuel integrity conditions were known.

The intention of the personnel in the TSC was to prevent damage to additional plant equipment, to protect the core, to minimize any release of radiation, to stringently adhere to procedures and to insure compliance with Technical Specifications.

Reactor Coolant System pressure was held at 1000 psig prior to decreasing below 425 degrees F as required by the Controlling Procedure for Unit Shutdown. (Reference page 17 of 18.) The basis for this action was to ensure rigid compliance with the cooldown curve and to determine the correct shutdown Boron concentration. Verification of boron concentration from sample analysis also contributed to this hold. During this hold period, S/G B pressure continued to decrease in conjunction

with the cooldown of the Reactor Coolant system. TSC personnel were fully aware that the pressure in the Reactor Coolant system was higher than the pressure in the B S/G until backfill of the B S/G was initiated. This decision was made to prevent diluting the Boron concentration in the Reactor Coolant system and to adhere to the shutdown curve. Another hold point occurred during preparation for the performance of the Power Operated Relief Valve temperature overpressure protection calibration and during the calibration itself.

As planned, representatives from Duke Power Company and the NRC staff maintained open line communications using the "Red Phone". On several occasions, NRC personnel requested direct telephone discussions with the TSC Emergency Coordinator. Some of these instances occurred at critical times and interfered with the Emergency Coordinator's management of the incident recovery. In addition, there were times when NRC personnel diverted the attention of Crisis Management Center resources with questions regarding previous actions.

The post trip review identified several abnormalities associated with this trip. Valve SV-7, S/G Power Operated Relief Valve, opened below setpoint. Work Request 88568 was written by Performance personnel and submitted to investigate and repair valve SV-7. The work request was later voided because the switches were recalibrated by Instrumentation and Electrical personnel under the quarterly preventative maintenance Work Request 097158. No computer indication was received that valves SA-48, Main Steam 1C to Auxiliary Feedwater Pump #1 Isolation, and SA-49, Steam Supply to Auxiliary Feedwater Pump, opened. However, work was in progress by Instrumentation and Electrical personnel pursuant to Work Request 096677 to replace a blown fuse. Valve 1NV-2, Reactor Coolant Letdown Isolation to Regenerative Heat Exchanger, did not operate properly and Work Request 137959 was submitted by Operations personnel to repair the valve. There was a Control Rod Data B failure on the control rods Digital Rod Position Indication EIIS:AA!. Work Request 137961 was written and submitted by Operations personnel to investigate and repair the cause of the failure. Instrumentation and Electrical personnel discovered a defective card and replaced it.

The response of radiation monitors (EMFs) during this event was as expected with one exception. 1EMF25, Steam Line B Radiation Monitor, 1EMF33, Condensate Air Ejector Radiation Monitor, and 1EMF36(L), Unit Vent Low Range Radiation Monitor all alarmed. 1EMF34, Steam Generator Blowdown Radiation Monitor did not alarm and a change in content rate did not occur. Normally, 1EMF34 would have initiated an automatic Blowdown isolation; however, this did not occur. A sample from the Blowdown of all four S/Gs is automatically taken by 1EMF34. It was expected by Health Physics personnel that 1EMF34 would have alarmed at about the same time that 1EMF25 alarmed. Health Physics personnel discovered that the demineralized water flush flow to 1EMF34 was valved in during the tube rupture event. This flow would have diluted any activity reaching the radiation monitor. In addition, the possibility of the flows from the four S/Gs not being equally distributed is being evaluated by Health Physics and Instrumentation and Electrical personnel.

A review of the McGuire LERs for the previous 12 months did not reveal any previous Reactor Trips resulting from a degradation of nuclear safety because of a steam generator tube rupture; therefore, this event is not considered recurring.

The Post Reactor Trip Plant Response is classified as a Category B. Three of the six response indicators exceeded the preferred or expected range, yet are not a significant concern.

This event is reportable to the Nuclear Plant Reliability Data System (NPRDS).

There were no personnel injuries or radiation overexposures as a result of this event.

CORRECTIVE ACTIONS:

Immediate: Operations personnel implemented procedure AP/1/A/5500/10, NC System Leakage Within the Capacity of Both Nv Pumps - Case I - Steam Generator Tube Leakage.

Subsequent: 1) Operations personnel manually tripped the Reactor and implemented procedure AP/1/A/5500/01, Reactor Trip.

2) Operations personnel implemented procedure RP/0/A/5700/02, Alert.

3) Operations personnel implemented procedure

RP/0/A/5700/10, NRC Immediate Notification Requirements.

4) A Steam Generator Tube Rupture Recovery Group was formed to determine the cause of the tube rupture and to develop and implement a recovery program.

5) A Technical Review Committee comprised of representatives from Westinghouse, Babcock and Wilcox, Dominion Engineering, Commonwealth Edison,and an independent NDE consultant was established to independently review the technical findings and recovery program.

6) Visual inspections were conducted to determine the location of the defective tube and to characterize the defect.

7) Full length Bobbin Coil Eddy Current Testing of 100% of the tubes was performed in A, B, C and D steam generators.

8) The defective tube and two additional tubes were removed.

9) Procedure OP/1,2/A/6100/02, Controlling Procedure for Unit Shutdown, and procedure OP/0/A/6100/06, Reactivity Balance Calculation, have been revised to more clearly allow cooldown initiation prior to meeting the Shutdown Margin for Cold Shutdown as long as the Shutdown Margin is maintained throughout the cooldown.

Planned: 1) Metallurgical analysis will be performed on the defective tube.

2) A final report will be issued by the Steam Generator Tube Rupture Recovery Group documenting the Steam Generator Tube Rupture recovery program and the results of any analysis performed on the defect.

3) Procedure AP/1/A/5500/10, NC System Leakage Within the Capacity of Both NV Pumps, will be evaluated for proper guidance throughout the steam generator tube rupture transient and appropriate changes will be made.

4) Procedure EP/1,2/A/5000/04, Steam Generator Tube Rupture, procedure EP/1,2/A/5000/4.2, Steam Generator Tube Rupture Cooldown Using Backfill,procedure EP/1,2/A/5000/4.3, Steam

Generator Tube Rupture Cooldown Using Blowdown and procedure AP/1,2/A/5500/01, Reactor Trip will be evaluated and enhancements will be made as appropriate. The changes will

be implemented after simulator validation is complete.

5) Procedure AP/1,2/A/5500/01, Reactor Trip, will be incorporated into procedure EP/1,2/A/5000/01, Safety Injection.

6) Full length bobbin coil eddy current testing will be conducted in all four steam generators on Unit 2 during the next Refueling Outage.

SAFETY ANALYSIS:

A steam generator tube leak is bounded by the analysis of a Steam Generator Tube Failure as described in Section 15.6.2 of the Final Safety Analysis Report (FSAR) Accident Analysis. The analyzed event is classified as an ANS Condition IV event, a limiting fault. Although a steam generator tube rupture is not an unforseen failure mode, it is considered a fault which is not expected to occur, but is postulated because of consequences with a potential for the release of significant amounts of radioactive material.

Reactor Coolant Activity

Technical Specification (TS) 3.4.8 specifies that the specific activity of the reactor coolant shall be limited to
Equivalent (DE) Iodine-131. Radiochemistry data obtained on March 6, 1989 at 0900 hours indicated Iodine-131 DE to be 0.025 microCuries/gram. By 0204 hours on March 8,1989, the value had increased to 0.5 microCuries/gram because of the typical activity spike which occurs as a result of a Reactor Trip. (Reference page 18 of 18.) TS 3.4.8 also requires that specific activity remains
average energy per disintegration. The value of E as determined on January 23, 1989 was 1.2 MeV/disintegration. The gross specific activity determined prior to the event was 1.6 microCuries/gm and is therefore considerably less than 100/E. These values for the limits on specific activity ensure that the resulting 2 hour radiation doses at the site boundary will not exceed an appropriately small fraction of 10CFR Part 100 dose guideline values

following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm.

The accident analysis for the S/G tube rupture event assumes 1% defective or failed fuel coincident with steam generator leakage of 1.0 gpm prior to the postulated accident for a time sufficient for specific activities to reach primary/secondary equilibrium. A comparison was made of isotopic activities obtained prior to and following initiation of the tube leak with those activity values specified in Table 15.0.9-2 of the FSAR for equilibrium conditions and 1% failed fuel. All of the sample values were found to be considerably less than the accident analysis.

Radionuclide Release Path

The accident analysis for a steam generator tube rupture event assumes that discharge of activity to the atmosphere takes place through the S/G safety and/or Power Operated Relief Valves. This path is considered because a coincident loss-of offsite power is also assumed which would result in a loss of main steam dumping capability. In the actual event, offsite power remained available, and therefore steam dump capability to the condenser EIIS:COND! was available and used. The activity transferred in the steam of S/G B was eventually released through the unit vent by way of the condensate air ejectors. Accordingly, EMF alarms were actuated as activity progressed from the S/G B steamline, to the condensate air ejectors, and then to the unit vent. Planned releases were also made as a result of the necessity to blowdown S/G B as part of the recovery procedure. A portion of the environmental release was through the assumed accident analysis flowpath as 3 of 4 S/G Power Operated Relief Valves opened for a duration of approximately three minutes. The Power Operated Relief Valve of the B S/G was isolated for maintenance, but is of no consequence from a safety point of view since the accident analysis assumes that the steam generators are controlled at the code safety valve open setpoint rather than the Power Operated Relief Valve open setpoint.

Control Room Operator Response and Safety System Performance

Upon immediate recognition of the situation as a steam generator tube rupture and leak, Control Room Operators implemented procedure AP/1/A/5500/10, NC System Leakage Within the Capacity of Both NV Pumps - Case I - Steam Generator Tube Leakage. The goal of the procedure is to lead the Control Room Operator through identification and isolation of the affected steam generator, cooldown of the NC system, equalization of primary and secondary pressure, and continued cooldown and shutdown of the unit. The accident analysis assumes that "the Operator identifies the accident type and terminates steam relief from the faulted steam generator within 30 minutes of the accident initiation". Additional response time is assumed to be available to the Control Room Operator for the more realistic cases of break sizes smaller than complete severance of a tube. The Control Room Operators immediately identified the accident type and the B Main Steam and Main Steam Bypass Valves were isolated within 11 minutes of the accident initiation, well within the expected 30 minute time frame. From a Control Room indication point of view, primary/secondary pressure equalization was accomplished according to the controlling procedure at approximately 45 minutes into the event.

The accident analysis has examined a complete severance of a double-ended rupture of a single S/G tube. Based on a S/G tube inside diameter of 0.67 inches, the leak flow area can be calculated to be 0.705 square inches with consideration given to a 2 source leak created in accordance with the double-ended break concept. With the actual crack observed to be generally of a narrow, elongated diamond shape in the tube's axial direction, a similar geometric figure with dimensions corresponding to those of the crack can be modeled for the purpose of comparing leak flow area. By interpreting the 'crack' model as four composite right triangles, an area of 0.349 square inches